NON-PUBLIC?: N
ACCESSION #: 9010010186
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 10

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP DUE TO THE FAILURE OF A FEEDWATER FLOW CONTROL VALVE
LINKAGE ARM NUT
EVENT DATE: 09/25/90 LER #: 90-025-00 REPORT DATE: 09/24/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 097

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: G. P. McGEE SUPERVISOR, COMPLIANCE TELEPHONE: (817) 897-5477

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: FCV MANUFACTURER: C635
X AB 52 W120
REPORTABLE NPRDS: YES
YES

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At approximately 0037 on August 25, 1990, Steam Generator (SG) Number 2 Feedwater Flow Control Valve (FCV) failed full open. A Reactor Operator attempted to close the valve from the Main Control Board by reducing the demand signal; however, at 0038, a turbine trip signal and feedwater isolation signal was generated due to Protection System Interlock P-14, Hi-Hi Level in SG Number 2. The reactor tripped at 0038 due to the turbine trip since reactor power was above the P-9 setpoint of 50 percent.

The cause of the Feedwater FCV failure is attributed to the feedback linkage arm from the valve stem to the valve positioner separating due to flow induced oscillations. Corrective actions include the installation

of a lock washer on the feedback linkage arm and a design modification to modify the valve internals to reduce flow induced oscillations.

END OF ABSTRACT

1. DESCRIPTION OF THE REPORTABLE EVENT

A. PLANT OPERATING CONDITIONS PRIOR TO THE EVENT

On August 25, 1990 at 0020, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, at 97 percent power.

B. REPORTABLE EVENT CLASSIFICATION

An event or condition that resulted in the manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)(EIIS:(JC)).

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

Not applicable - no structures, systems or components were inoperable at the start of the event that contributed to the event.

D. NARRATIVE SUMMARY OF THE EVENT, INCLUDING DATES AND APPROXIMATE TIMES

On August 25, 1990 at 0023, a transient began in the Heater Drain System (EIIS:(SN)) due to a failure of a level control valve (EIIS:(LCV)(SN)). As a result of the transient, the feedwater temperature decreased, which resulted in an increase in reactor power. At 0024, the Reactor Operator (RO) (utility, licensed) began to reduce turbine (EIIS:(TRB)(TA)) load to maintain reactor power below 100 percent. At 0037, Steam Generator (SG) Number (No.) 2 (EIIS:(SG)(AB)) Feedwater Flow Control Valve (FCV) (EIIS:(FCV)(SJ)) failed full open. The RO attempted to close the valve from the Main Control Board (EIIS:(MCBD)(IB)) by reducing the demand signal; however, at 0038, a turbine trip signal and feedwater isolation signal was generated due to Protection System Interlock P-14, Hi-Hi Level, in SG No. 2. The reactor (EIIS:(RCT)(AB)) tripped at 0038 due to the turbine trip since reactor power was above the P-9 setpoint of 50 percent.

After the reactor trip, several SG Lo-Lo Level reactor trip signals and auxiliary feedwater actuation signals were received from SGs 1, 3, and 4 due to instrumentation channels spiking low on the sudden changes in main steamline pressure. The phenomenon of SG level channels spiking low on changes in main steamline pressure is discussed in detail in LER 90-021-00.

The auxiliary feedwater actuation signal caused an automatic start of the Motor Driven Auxiliary Feedwater Pumps (EIIS:(P)(BA)). The Turbine Driven Auxiliary Feedwater Pump (TDAFWP) (EIIS:(P)(BA)) did not start.

During the transient, prior to the reactor trip, Pressurizer Backup Heater Group A (EIIS:(EHTR)(AB)) failed to energize due to a breaker(EIIS:(52)(AB)) problem. Pressurizer (EIIS:(PZR)(AB)) pressure, however, never went below the low pressure reactor trip point. Pressure was restored within 15 minutes of the reactor trip.

An event or condition that results in a manual or automatic actuation of any ESF, including the RPS, is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At 0154 on August 25, 1990, the Nuclear Regulatory Commission Operations Center was notified of the event via the Emergency Notification System. The plant was stable and in Mode 3.

E. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM FAILURE OR
PROCEDURAL ERROR

Failure of the SG No. 2 Feedwater FCV and Pressurizer Backup Heater Breaker were discovered during the event by Control Room (EIIS:(NA)) indications. The TDAFWP not actuating on Lo-Lo Level signals from two SGs was noted during the event.

II. COMPONENT OR SYSTEM FAILURES

A. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED

COMPONENT

1. Feedwater FCV

The SG No. 2 Feedwater FCV failed full open due to the feedback linkage separating from valve's positioner (EIIS:(33)(SJ)). The full open valve overfed SG No. 2 causing a turbine trip and reactor trip. The valve successfully closed on the feedwater isolation signal associated with the trip.

2. Pressurizer Backup Heater

The Pressurizer Backup Heater Group A power supply breaker failed to close on demand due to failure of the breaker close coil. The breaker is designed to close on a low pressurizer pressure signal to energize the Pressurizer Backup Heaters Group A. Pressurizer pressure control was maintained by the other three Pressurizer Heater Groups which remained operable. Failure of the breaker did not contribute to the resulting reactor trip.

3. TDAFWP

The TDAFWP requires Lo-Lo Levels in more than one SG for an actuation. Simultaneous Lo-Lo Levels in two SGs occurred twice after the trip; once for 0.23 seconds and once for 0.039 seconds. Neither start signal was long enough to cause the steam supply valves (EIIS:(V)(BA)) for the TDAFWP to open far enough to clear their closed limit switch, and lock in the valve open signal. The TDAFWP, therefore, did not start. The pump was available to respond to a sustained low SG level conditions. Subsequent testing showed the SG Lo-Lo Level signal from two or more SGs must be present for at least one second before the TDAFWP steam admission valves stroke far enough to clear their closed limit switch and lock in the pump start.

B. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE

1. Feedwater FCV

The immediate cause of the SG No. 2 Feedwater FCV failing open was the valve stem position feedback linkage arm to the valve positioner separating due to the nut becoming loose. The positioner drive arm was then driven by the positioning spring to the valve closed position. The positioner responded to the erroneous closure signal by porting full output pressure to the valve diaphragm, causing the valve to go full open. After receiving the SG No. 2 Hi-Hi Level signal, which resulted in a feedwater isolation signal, the redundant solenoid valves (EIIS:(FSV)(SJ)) between the positioner and the valve diaphragm de-energized to vent the diaphragm and close the valve.

The mechanism by which the feedback linkage arm nut became loose is uncertain but is believed that flow induced valve oscillations allowed the mounting nut to loosen slightly. The nut backed off the cap screw, allowing the drive rod to separate from the drive arm.

Subsequent investigation following an additional failure of this valve on September 7 showed a worn area on the positioning cam, a groove worn into the cam roller, and binding in the feedback drive rod ball and socket joints. These problems did not contribute to the August 25 failure of the valve.

2. Pressurizer Backup Heater Breaker

The immediate cause of the failure of the Pressurizer Backup Heater breaker to close was a failure of the breaker close coil. TU Electric is continuing to investigate the cause of the breaker failure. This breaker has a relatively high service demand with frequent cycling.

C. SYSTEMS OR SECONDARY FUNCTIONS THAT WERE AFFECTED BY FAILURE OF
COMPONENTS WITH MULTIPLE FUNCTIONS

Not applicable - there were no failed components with multiple functions involved.

D. FAILED COMPONENT INFORMATION

1. Feedwater FCV

Actuator
Tag Number: 1-FCV-0520A0
Manufacturer: Copes-Vulcan Division
Model Number: D-100-160

Positioner
Manufacturer: Bailey Controls
Model Number: 5221030-8

2. Pressurizer Backup Heater Breaker
Tag Number: 1 PCPR1
Manufacturer: Westinghouse
Model Number: DS416

III. ANALYSIS OF THE EVENT

A. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety systems actuated automatically as a result of the event. The appropriate components within these systems operated as designed upon receipt of the Steam Generator Hi-Hi Level signal.

Feedwater Isolation Valves (EIIS:(ISU)(SJ))
RPS
Turbine Generator Trip System (EIIS:(TG))
Motor Driven Auxiliary Feedwater Pumps

B. DURATION OF SAFETY SYSTEM TRAIN INOPERABILITY

Not applicable - there were no safety systems which were rendered inoperable due to this failure.

C. SAFETY CONSEQUENCES AND IMPLICATIONS OF THE EVENT

The reactor trip on August 25 occurred while a transient on a plant secondary system was in progress. Based on a review of transient data, the effects of the failure of the Feedwater FCV to close which resulted in the turbine trip and subsequent reactor trip was overshadowed by the secondary system transient which caused a lower feedwater temperature and an increase in reactor power prior to the trip. The event on August 25 is bounded by the "Excessive Increase In Secondary Steam Flow" transient analysis presented in CPSES Final Safety Analysis Report (FSAR) 15.1.3. An excessive increase in the secondary system steam flow is defined as a rapid increase in steam flow that causes a power mismatch between reactor core power and the SG load demand. A comparison of the values of parameters predicted by the accident analysis and the values observed during the August 25 event is presented below.

August 25
FSAR Event

Vessel Average 15 degrees 12 degrees F.

Temperature Decrease Fahrenheit (F)

Final Pressurizer Pressure 2030 psia 2200 psia

Reactor Power Increase from Increased
(prior to trip) 100 percent to from 97
105 percent percent to
105 percent

Core Flow Rate Thermal Design Best-Estimate
Flow (Approximately
95 percent of
Best-Estimate)

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The accident presented in the FSAR is analyzed to show that the minimum Departure from Nucleate Boiling (DNB) ratio does not fall below the design limit. As shown in FSAR 15.1.3, the DNB ratio does not fall below its initial value during the accident. For the August 25 event, the vessel average temperature change was lower than was assumed in the accident analysis, the pressurizer pressure higher, and the core flow higher. The direction of change of each of these parameters relative to the FSAR analysis would result in a relatively higher DNB ratio. In addition, the Over Temperature N-16 System (OTN-16) is designed to trip the reactor to prevent DNB from occurring. Because the OTN-16 trip setpoint was not reached during the transient, no DNB occurred and the accident analysis acceptance criteria continued to be met.

Based on the smaller change in vessel average temperature and on the evidence that the minimum DNB ratio is greater than calculated for the accident analysis, the August 25 event is bounded by the "Excessive Increase in Secondary Steam Flow" accident analysis. Therefore, the event did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

IV. CAUSE OF THE EVENT

The cause of the event is attributed to failure of SG No. 2 Feedwater FCV. The mechanism of the failure is discussed in Section II.

V. CORRECTIVE ACTIONS

A. IMMEDIATE

The SG No. 2 Feedwater FCV feedback linkage drive rod was reconnected to the positioner drive arm with a nut and lock-washer. The other SG Feedwater FCVs were inspected for loose parts; no loose parts were discovered.

The Pressurizer Backup Heater Group A breaker was replaced.

B. CORRECTIVE ACTIONS TO PREVENT RECURRENCE

Root Cause

Failure of the feedback arm was due to flow induced valve oscillations.

Corrective Action

1. The positioner has been replaced and a lock-washer was installed with the linkage arm nut.

2. A design modification has been approved which will modify the valve internals to reduce flow induced oscillation of the Feedwater FCV. As a temporary measure, the valve has been repacked to reduce oscillation.

3. A preventive maintenance activity has been added to inspect and lubricate the Feedwater FCV positioner at power on a monthly basis.

C. CORRECTIVE ACTION TAKEN ON GENERIC CONCERNS IDENTIFIED AS A

DIRECT RESULT OF THE EVENT

Generic Considerations-1

A similar failure may occur in the other three Feedwater FCVs.

Corrective Action-1

Corrective Actions to the Root Cause identified for SG No. 2 FCV will be implemented on the other three valves.

Generic Considerations-2

The possibility exists that the air operated control valves with Bailey positioners in other systems may be subject to flow induced oscillations.

Corrective Action-2

Air operated control valves with Bailey positioners are being inspected for similar positioner problems. The Bailey positioners in the plant primary systems do not perform a safety-related function. No problems to date have been identified. Any identified problems will be addressed on a case by case basis.

VI. PREVIOUS SIMILAR EVENTS

Although there have been several previous events (LER 90-002, LER 90-009, LER 90- 013, LER 90-017, LER 90-23) that resulted in reactor trips, the root causes of those events were unrelated to the root cause of this event. The corrective actions taken to resolve the root causes of the previous events would not have prevented this event. Therefore, no previous similar events have been reported pursuant to 10CFR50.73.

VII. ADDITIONAL INFORMATION

The times listed in the report are approximate and Central Daylight Savings Time (CDT).

TUELECTRIC

Austin B. Scott, Jr. CPSES-9021654
Vice President September 21, 1990

No Response Required

TO: J. W. Beck - ST 24

SUBJECT: LICENSEE EVENT REPORT 50-445/90-025-00
REACTOR TRIP DUE TO FAILURE OF FEEDWATER
FLOW CONTROL VALVE FEEDBACK LINKAGE NUT

Attached is Licensee Event Report (LER) 50-445/90-025-00 which has been prepared in accordance with 10CFR50.73(d). This LER has been reviewed by SORC (Meeting No. 90-159) and recommended for approval. Additionally, I have reviewed and approved the LER and find it acceptable for submittal to the NRC (required by September 24, 1990).

If you should have any questions, please contact Gary McGee at extension 5477.

A. B. Scott, Jr. O09
JCG:aki

Attachment

cc: CCS E06 OL,1A
W. G. Guldemond CL1 1L,1A
R. D. Walker ST-24 1L, 1A

P. O. Box 2300 Glen Rose, Texas 76043-9990 817-897-5672

ATTACHMENT 1 TO 9010010186 PAGE 2 OF 2

Log # TXX-90312
File # 10200
910.4
Ref. # 50.73(a)(2)(iv)

TUELECTRIC

William J. Cahill, Jr.
Executive Vice President September 24, 1990

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk

Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEERED SAFETY FEATURE
LICENSEE EVENT REPORT 90-025-00

Gentlemen:

Enclosed is Licensee Event Report 90-025-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Resulting From Failure of Feedwater Flow Control Valve Feedback Linkage."

Sincerely,

William J. Cahill, Jr.

KWV/daj

Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

400 North Olive Street L.B. 81 Dallas, Texas 75201

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